CP&L
Carolina Power & Light Company

June 14, 1982

Mr. Darrell G. Eisenhut, Director
Division of Licensing
United States Nuclear Regulatory Commission
Washington, D.C. 20555

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324
LICENSE NOS. DPR-71 AND DPR-62
GENERIC LETTER NO. 81-34
BWR SCRAM SYSTEM PIPING INTEGRITY

Dear Mr. Eisenhut:

SUMMARY

Carolina Power & Light Company (CP&L) has reviewed NUREG-0803 entitled "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping", which was transmitted by Generic Letter 81-34 dated August 31, 1981. In general, we agree with the results of the analyses; however, we believe that the inclusion of specific data, design features, and code requirements appropriate for Brunswick Steam Electric Plant (BSEP) results in significantly lower probabilities of occurrence of the postulated failures and shows that multiple options for mitigating the consequences are available. Therefore, the conclusions reached in the generic analysis are overly conservative and not justified on the basis of risk to the public.

DISCUSSION

A three-phase approach was used by NUREG-0803 to evaluate the safety concerns regarding the integrity of BWR Scram System piping. The first step evaluated the probability of occurrence of Scram Discharge Volume (SDV) System piping failure. The NRC risk analysis determined a frequency of core melt less than 1×10^{-6} per year and concluded that the analyzed sequence of events is not a dominant contributor to core melt.

The risk analysis performed by General Electric (GE) determined a core melt frequency of less than 2×10^{-9} per year, significantly less than the NRC estimate. It appears the difference in results arises primarily from two probability estimates:

1. SDV Pipe Break Probability

   NUREG-0803 conservatively overestimated the amount of SDV piping present and, therefore, the probability of pipe failure by a factor

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of two. In addition, we do not believe sufficient credit was taken for the minimal amount of time the piping is stressed, the absence of mechanisms which support crack growth, and the crack before break characteristics of the piping. These factors justify a significant reduction in the probability of failure of the SDV piping over that for high-energy piping which is continually stressed and exposed to normal crack growth mechanisms.

The BWR Owners' Group is in the process of evaluating and revising the generic probabilistic risk assessment of scram system piping failures (NEDO-24342) prepared by GE. The results of the revised assessment will be submitted to the NRC upon completion.

2. Inability to Depressurize Capability

The predominant contributor to a failure to depressurize was determined by NUREG-0803 to be operator error. The guidance to be contained in the Emergency Procedure Guidelines will provide specific operator instructions on the diagnosis and subsequent actions to take in response to a break outside containment; therefore, the probability of operator error is minimal. Also, the operating steps that must be taken to depressurize are straightforward and familiar to the operators. The safety/relief valves which could be used to depressurize the reactor have been shown by operating experience to be highly reliable. Located in the drywell, they would be shielded from any environmental effects of the SDV break. The redundancy and independency offered by the eleven valves represent a highly reliable method of depressurization. In addition, operation of the High Pressure Coolant Injection (HPCI) System offers another potential method of pressure reduction not considered. The need to depressurize would only arise out of an inability to reset the scram signal. The NUREG determined that methods exist for clearing all but two scram signals - drywell high pressure and main steam line high radiation. Drywell high pressure could only be caused by a pipe break inside the drywell or by accidental drywell overpressurization. The former is not a credible event when combined with another break outside containment. The latter can be quickly cleared by venting the drywell through the Standby Gas Treatment (SGBT) System. The inability to reset a high drywell pressure scram is, therefore, unlikely.

The main steam line radiation monitor response is due primarily to N-16. Once a scram occurs, the main steam line monitor radiation levels would quickly decrease due to the extremely short half-life of N-16 and the closure of the Main Steam Isolation Valves (MSIVs), stopping flow past the monitors. The scram could then be reset. This response has been demonstrated at BSEP when high radiation from a resin injection initiated MSIV closure.

Therefore, for all scram signals, it is unlikely that the scram could not be reset within a short period of time, terminating

leakage through any break. One of the early actions taken following a scram is to reset it as soon as possible. The operator will, therefore, take the best mitigating action to isolate the break without ever having to diagnose that a break has occurred on the SDV piping.

We believe that these two factors together result in an overestimate of the risk by the NRC by a factor of 100, that a probability of less than 1×10^{-8} is more appropriate for BSEP, and that the probability of less than 2×10^{-9} determined by GE is realistic. These levels of risk are acceptable and, therefore, no further evaluation is necessary.

NRC Standard Review Plan 3.6.1 provides rules for pipe rupture evaluation. It requires breaks to be postulated for high-energy piping and cracks to be postulated for moderate energy piping whenever the piping is normally pressurized. Since the SDV piping is not pressurized, but vented during normal operation and is only pressurized for short periods following a scram (i.e., an upset condition), no pipe break or even crack need be postulated per SRP 3.6.1 criteria.

The second phase of evaluation considered in NUREG-0803 reviewed the consequences of the failure of the SDV piping on plant equipment and the potential for affecting essential safety functions. Although the low probability of failure of the SDV piping and the provisions of SRP 3.6.1 precludes the need to perform the second phase of the evaluation, it was reviewed for additional insight. Since BSEP has 137 control rod drives (CRDs), the maximum leak rate is estimated to be 411 gpm. Due to the leak before break characteristics of the piping, the actual leakage would be expected to be much less than this.

The effects of a break in the SDV piping on equipment needed to mitigate the consequences of the break are minimized by the physical separation and independence of the Emergency Core Cooling System (ECCS) subsystems and the number of systems available for core cooling. Building drains, walls, curbs, and overhangs protect ECCS equipment from significant impingement by water. In addition, any such impingement would be cooled significantly through contact with floors, walls, and stairways to well below 212°F. Due to the elevation of equipment off the floor, the ECCS equipment would not be affected by flooding from a SDV pipe break during the transient and, even if it were, the effects would be delayed and limited to only a portion of the equipment since the subsystems are physically separated, leaving sufficient independent subsystems operable to achieve core cooling.

The ability to maintain core cooling is not dependent upon the availability of the high-pressure makeup systems. As soon as it is recognized that core uncovery is likely, the operator response is to perform a manual depressurization of the reactor. Core cooling can then be provided by many independent, diverse systems, both inside and outside the Reactor Building.

Review of the possible impacts of a SDV pipe break have not determined any new failure mechanisms which have not already been reviewed by

previous bounding analysis or are not under current evaluation by IE Bulletin 79-01B activities. Sufficient systems are available to mitigate the consequences of the break and assure core cooling; therefore, the consequences of the break are acceptable.

The third phase evaluated the need for improved mitigation capability. The results of the analyses from Phase I and Phase II indicate that Phase III improvements are not needed to assure plant safety.

An individual item review of the summary of the Staff's guidance and schedule for resolution of the outstanding issues and concerns contained in Table 5.1 of NUREG-0803 follows:

1. Periodic In-Service Inspection

   Since the redundant seals and restricting flow areas in the CRDs isolate the withdraw lines from the reactor coolant pressure boundary, the SDV piping should not be considered as extensions of the reactor coolant pressure boundary penetrating containment. The SDV piping, therefore, does not meet the requirement for classification as Class 2 piping under the "Boiler and Pressure Vessel Code" of the American Society of Mechanical Engineers (ASME Code), in accordance with the criteria of Regulatory Guide 1.26, nor are these lines necessary for achieving the scram function. Therefore, there is no code basis for justifying in-service inspection (ISI) requirements on the SDV piping. Our evaluation of the probability of failure of the piping and of the consequences of the failure also indicate that there is no prudent basis for establishing ISI requirements. The installation of the system under a QA program and 100% NDE (radiography) of all welds during construction assures the integrity of the installation. The NDE included radiography on all piping welds greater than two inches in diameter. The lack of credible defect growth mechanisms and negligible service duty exclude any failure mechanism. The high radiation levels from the SDV piping also present an ALARA concern if there are no anticipated benefits gained from inclusion of this system in the ISI programs. Our current ISI program is in accordance with the requirements of the Winter 1977 Edition of the Section XI ASME Code with Addenda through Summer 1978. Reviews of the requirements of this edition of the code have not determined any basis for instituting ISI on the SDV piping.

2. Threaded Joint Integrity

   A review of the SDV System piping found one threaded connection on the B loop scram discharge instrument volume. This connection, a 3/4-inch - 6,000# threaded pipe cap, is on a 4-3/4-inch long spare tap located approximately half-way between the seismic supports on the 12-inch diameter instrument volume.

   Shock and vibration would not be detrimental to cap or nipple integrity due to design and location as described above. System

operation itself would restrict erosion and corrosion in the nipple and cap since it is a "no-flow" area, and the volume is only filled and pressurized less than 2% of the time. A visual inspection of the cap and nipple indicated good condition and integrity with no indication of leakage.

The structural and leaktight integrity of the threaded cap, would, therefore, not be decreased under any analyzed conditions. To still further assure the integrity of the threaded connection, the cap will be seal welded during the 1982 refueling outages.

3. Seismic Design Verification

A program to perform seismic reanalysis of the essential piping in the CRD System in accordance with the guidance contained in IE Bulletin 79-07 was recently completed for all essential CRD supports outside the drywell. The necessary modifications to restore the desired design margins are being designed and installed. Installation of the remaining modifications is scheduled for completion by the end of the 1982 refueling outages on each unit. The results of this program will be transmitted to NRC in a final report associated with the IE Bulletin 79-07 and IE Bulletin 79-14 requirements.

4. HCU-SDV Equipment Procedures Review

Work performed on safety-related equipment must be performed under the control of procedures approved by the Plant Nuclear Safety Committee. They provide assurance that maintenance and surveillance activities are performed in a safe manner. One such procedure, the clearance procedure, assures that the equipment being worked on remains isolated, drained, and vented for the duration of the work for personnel protection. Where positive isolation cannot be obtained, work is not permitted until plant conditions are correct. For HCU-SDV equipment, cold shutdown would be required. Current procedures require component isolation or specify that the plant be in cold shutdown condition in order to perform maintenance on the SDV System.

5. Environmental Qualification of Prompt Depressurization Function

The prompt depressurization function would be performed by the Automatic Depressurization System (ADS). This system has already been determined to require environmental qualification and has been included under the IE Bulletin 79-01B evaluation. That evaluation will bound any probable SDV line break and so is not expected to create any worse effects on the system. Therefore, no further evaluation is needed.

6. As-Built Inspection of SDV Piping and Supports

A program to perform as-built inspections of the piping and supports in the CRD System in accordance with the guidance contained in IE Bulletin 79-14 was recently completed. This information has been

factored into the seismic reanalysis effort under IE Bulletin 79-07. The results of this program will be transmitted to NRC in a final report associated with the IE Bulletin 79-07 and IE Bulletin 79-14 requirements.

7. Improvement of Procedures

Per its charter, the BWR Owners' Group cannot respond to NRC requests for utility action, except at the discretion of its members. Neither can CP&L commit the BWR Owners' Group to a specific course of action except by its participation in Owners' Group decisions by vote. Thus, CP&L can only provide a response to the Staff's guidance to the BWR Owners' Group in NUREG-0803 as if it were addressed to CP&L directly.

However, the BWR Owners' Group has discussed the guidance of NUREG-0803 regarding modification of the Emergency Procedure Guidelines and acknowledges the benefits of treating the subject generically. The BWR Owners' Group is in the process of completing an extension of the guidelines to include steps per reactivity control, and certain other modifications to the guidelines which have been discussed with your staff. It is CP&L's judgment that completion of these modifications outweighs, in immediate importance, the NUREG-0803 guidance for other guideline modifications. After current activities on the guidelines are substantially complete, CP&L will support a preliminary study by the BWR Owners' Group to determine the best approach to fulfilling the intent of the guidance provided in NUREG-0803. It is not clear that the best approach will involve modification of the guidelines. When that study is complete, the Owners' Group will determine whether to authorize specific actions to modify the Emergency Procedure Guidelines. CP&L will advise the NRC of the result of that decision and the Owners' Group plan at that time.

8. Verification of Equipment Designed for Water Impingement

A review was conducted of SDV System piping layout to determine what ECCS components could potentially be lost by water impingement from a rupture of the SDV System. As discussed earlier, building walls, curbs, and overhangs protect most ECCS equipment from impingement by water, thus reducing its effects significantly. All valve operators onto which water could be sprayed are NEMA 4 enclosures and, therefore, are not susceptible to spray. Due to the divisional and physical separation of the ECCS subsystems, a break would, at worst, result in system divisional losses and, therefore, leave adequate subsystems to achieve core cooling.

9. Verification of Equipment Qualified for Wetdown by 212°F Water

As stated above, a review was performed of SDV System piping layout and safety-related equipment locations to determine those components that could potentially be lost by wetdown from a break in the SDV

System. Due to component location, electrical separation, and predicted water temperatures of less than 212°F, it was found that sufficient core cooling equipment would remain operational to achieve core cooling.

10. Verification of Feedwater and Condensate System Operation Independent of the Reactor Building Environment

Operation of the Feedwater System is dependent upon Reactor Building equipment in two ways:

a. Steam for driving the reactor feed pumps is provided from the reactor via the MSIVs.

b. Reactor high-level instrumentation in the Reactor Building can trip the reactor feed pumps.

The instrumentation in the Reactor Buildings which affects the reactor feed pumps is on another elevation where analysis shows the predicted ambient temperatures would be less than 140°F. Even if the reactor feed pumps were unavailable, there are six pumps in the condensate and heater drain systems whose discharge pressures are greater than 350 psi. The capacity of these pumps is sufficient to maintain reactor level even without full depressurization.

11. Evaluation of Availability of HPCI-LPCI Turbines Due to High Ambient Temperature Trips

Analysis shows that the temperatures in the vicinity of the HPCI and RCIC high-temperature isolation switches remain well below their set point during any probable SDV pipe break. The operation of HPCI and RCIC would, therefore, not be impacted by such a break. The Low Pressure Coolant Injection (LPCI) System is a subsystem of the Residual Heat Removal (RHR) System and, therefore, uses the motor-driven RHR pumps for coolant injection under accident conditions.

12. Verification of Essential Components Qualified for Service at 212° F and 100% Humidity

Qualification of system components required for the safe shutdown of the reactor is already included under the IE Bulletin 79-01B evaluations. High-energy line break analyses required by IE Bulletin 79-01B will bound any probable SDV line break, particularly in view of its leak before break chacteristics. Routine walk-throughs of this area by auxiliary operators each shift would identify any evidence of leakage promptly before failure could progress to a rupture.

13. Limitation of Coolant Iodine Concentration to a Standard Technical Specification Values

BSEP has implemented the Standard Technical Specifications (STS).

In summary, CP&L believes that the inclusion of plant specific data and more representative assumptions into the evaluation of the impact of a break in the SDV piping causes the probability of occurrence of such a failure to become acceptably small, that the applicable codes and NRC review criteria do not require that a break be postulated, and that the impact on mitigating equipment is bounded by the effects of breaks being considered by IE Bulletin 79-01B analysis. As a result, the planned actions are sufficient since the postulated failure is not a dominant contributor to core melt.

Yours very truly,

P. W. Howe
Vice President
Technical Services

WRM/JSB/lr (n-79)

cc: Mr. J. P. O'Reilly (NRC-RII)
Mr. D. B. Vassallo (NRC)
Mr. J. Van Vliet (NRC)

P. W. Howe, having been first duly sworn, did depose and say that the information contained herein is true and correct to his own personal knowledge or based upon information and belief.

Franklin Murray
Notary (Seal)

FRANKLIN MURRAY
NOTARY PUBLIC
WAKE COUNTY, N. C.

My commission expires: October 4, 1986